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REACTOR OPERATIONS
AND RADIOACTIVE WASTE OPERATIONS
QUARTERLY REPORT

April - June, 1959

By

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OAK RIDGE NATIONAL LABORATORY
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REACTOR OPERATIONS

AND RADIOACTIVE WASTE OPERATIONS

QUARTERLY REPORT

Summary

The number of unscheduled shutdowns of the ORR increased to eighteen in the quarter. Most of the shutdowns were due to failures in the reactor controls and to electrical storms which interrupted the electrical power. Modification of the ORR coolers was completed by the manufacturer, but tests showed them to be approximately 24% deficient in capacity. This makes it impossible to operate at 20 Mw on hot days. Engineering has begun on a new cooling system.

The proposal for annealing the Graphite Reactor was submitted for safeguard approval on April 20, but it does not appear that approval will be received in time to perform the anneal this summer. Since hot weather is required, it will be necessary to do this operation in the summer of 1960. Production fuel slugs can apparently be used in place of the regular Graphite Reactor slugs. This may eliminate the necessity of producing more slugs especially for the Graphite Reactor.

The Process Waste Treatment Plant removed 92% of a spill of Pm^{147} and 70% of the Sr^{90}.

1. OAK RIDGE RESEARCH REACTOR

1.1 Operations

W. R. Casto

Operations

The operations data for the ORR are given in Table 1.1. Total operating time for this quarter was 1573.352 hr.

TABLE 1.1 ORR OPERATIONS

Period April 1, 1959 through June 30, 1959

	This Quarter	Last Quarter	Year to Date
Total energy, Mwd	1261.2	1315.4	2576.6
Average power, Mw-operating hour	19.2	19.1	19.2
Time operating, %	72.0	76.4	74.2
Reactor water radioactivity, c/m/ml (av)	30,348	28,375	29,361
Pool water radioactivity, c/m/ml (av)	593	670	631
Reactor water resistivity, ohm-cm (av)	880,000	780,000	830,000
Pool Water Resistivity, ohm-cm (av)	855,000	950,000	902,000
Research samples	20	12	32
Radioisotope samples	96	63	159

A change in core configuration, Figure 1.1, was made to provide the optimum thermal neutron flux for the experiment in HN-1.

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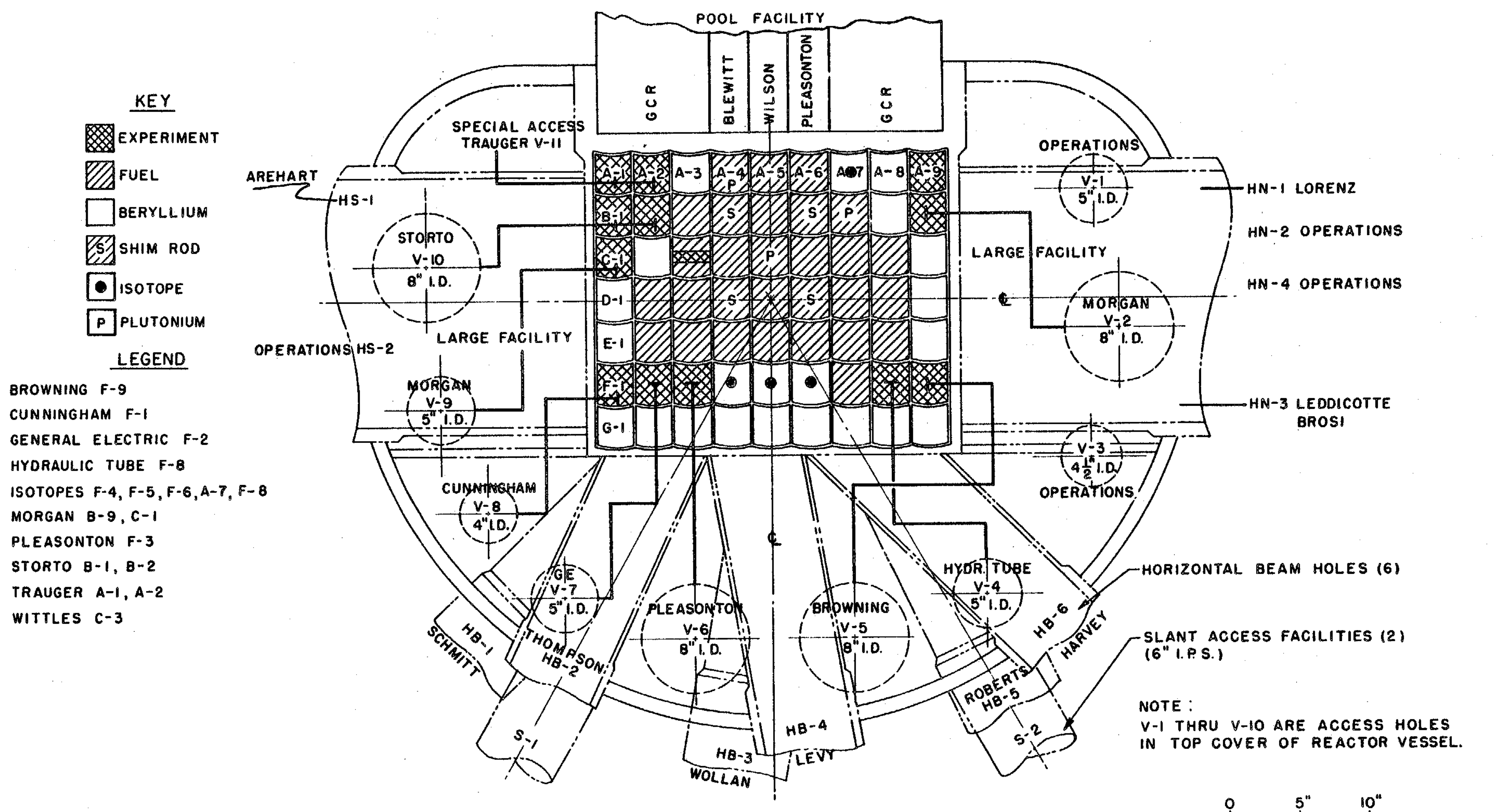

Fig. 1.1.
O.R.N.L. RESEARCH REACTOR
LATTICE PATTERN AND EXPERIMENT LOCATIONS

RESEARCH ASSIGNMENTS &
FUEL LOADING ON 6-30-59

Cycles of operation during this period are shown in Table 1.2.

TABLE 1.2. CYCLES OF OPERATION

Cycle No.	Date Begun	Date Ended	Accumulated Energy (Mwd)
10	April 11	April 19	155.7
11	April 23	May 17	457.8
12	May 22	June 16	466.2
13	June 19	In Progress	181.5

Figures 1.2, 1.3 and 1.4 indicate shim rod positions vs. operating time of cycle during cycles 10, 11, and 12, respectively.

<u>Radioactivity Release</u>

At the beginning of cycle No. 11 (4-23-59), significant amounts of air radioactivity inside the ORR building were encountered. The reactor was shut down to remove an experiment from position B-9 which had been inserted prior to the beginning of Cycle No. 11. Following the start-up, the building background was normal.

Inspection and testing of the experiment indicated a faulty kovar seal which permitted radioactive gases to escape into the building.

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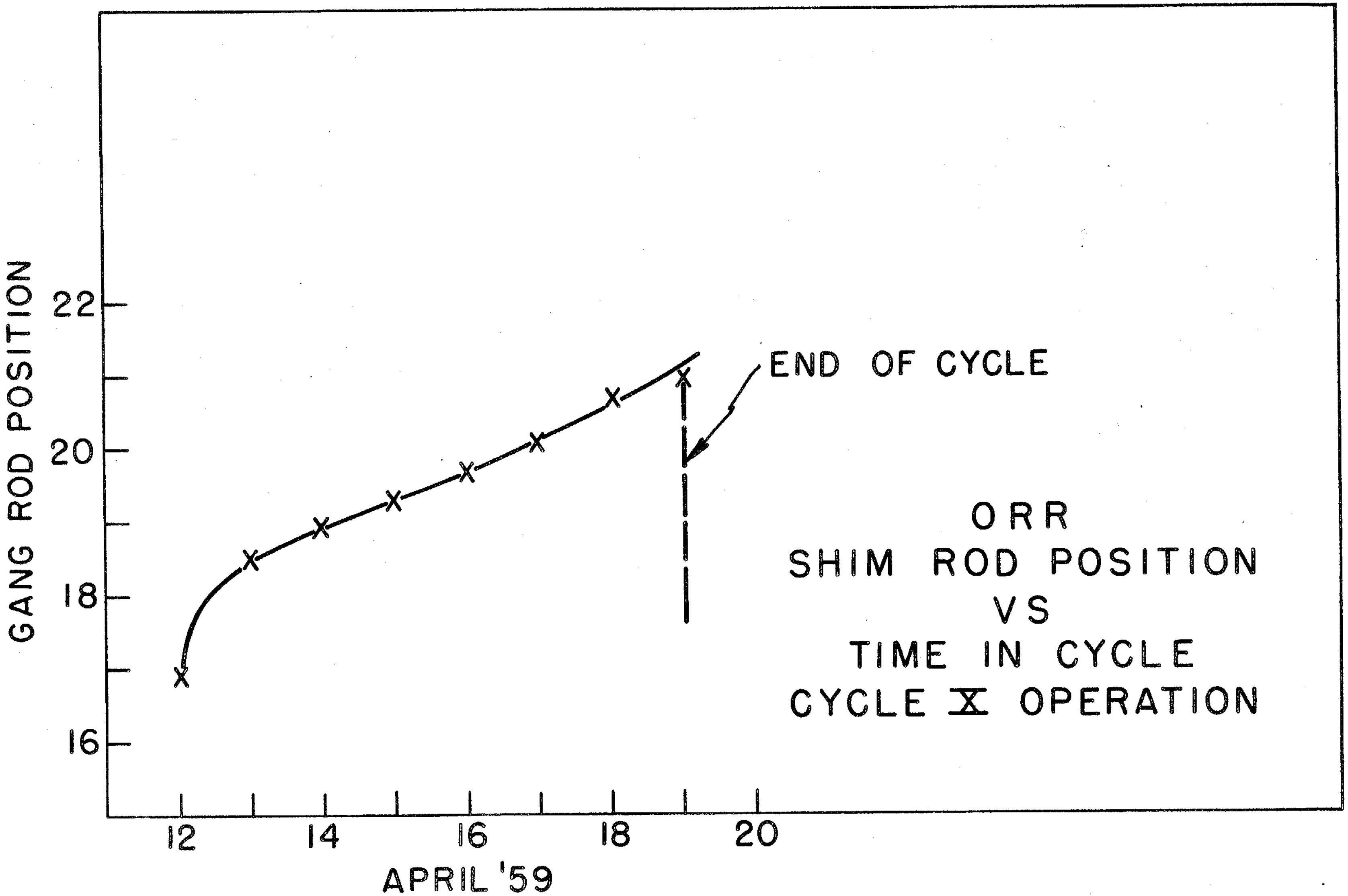

Fig. 1.2

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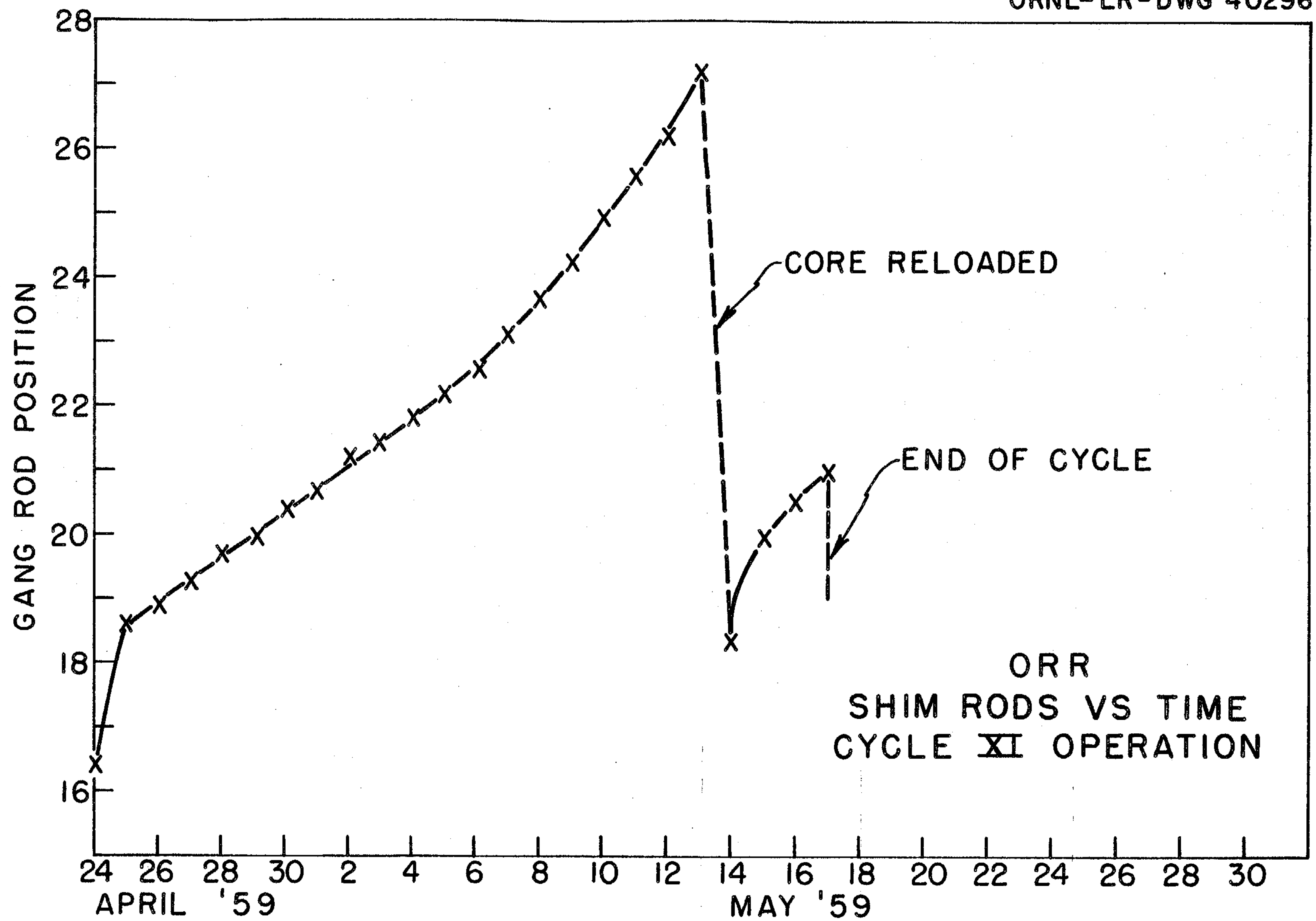

Fig. 1.3.

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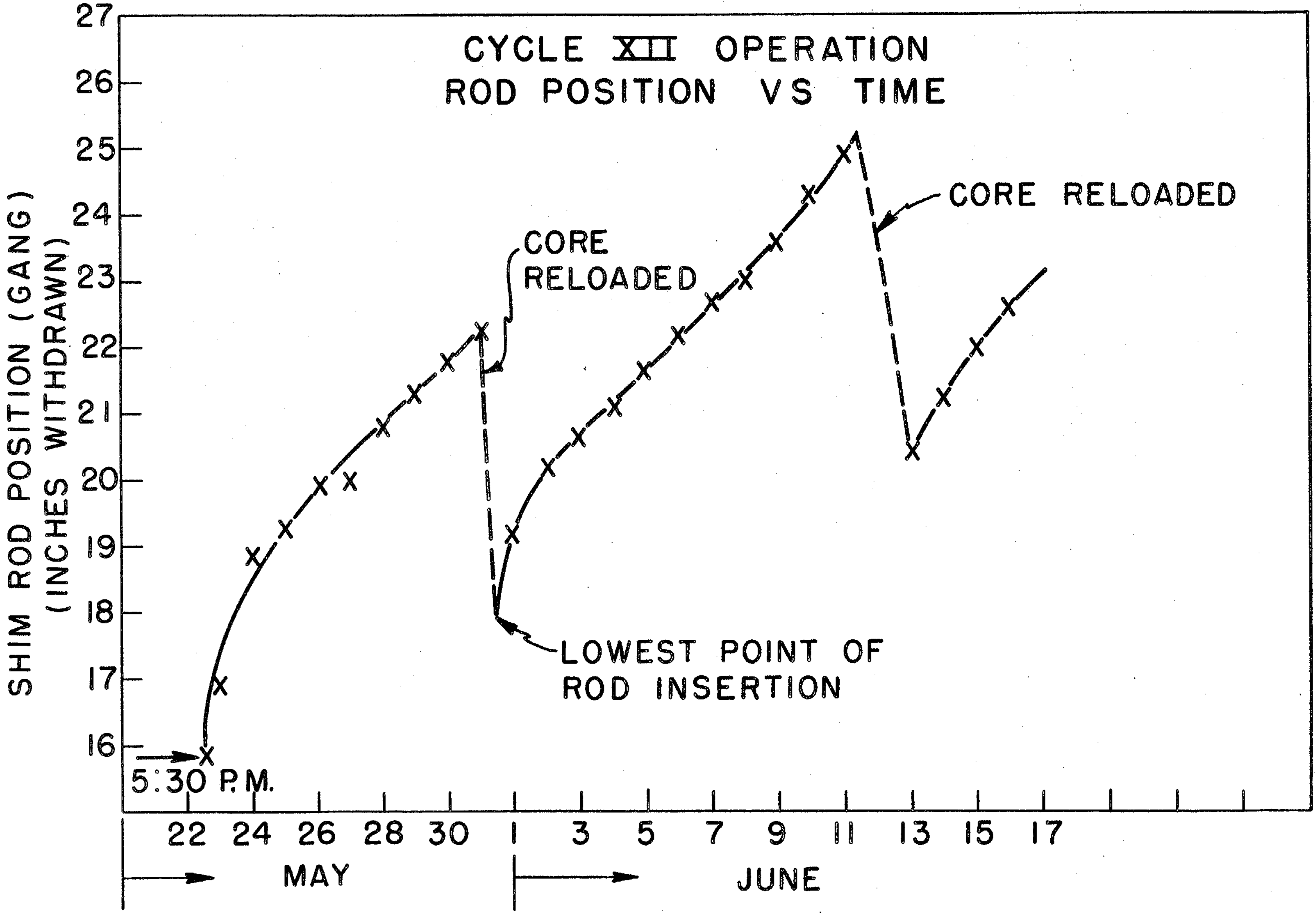

Fig.1.4.

Shutdowns

Three major shutdowns occurred during this period.

April 19 through April 23. Flux measurements were made to determine the optimum fuel arrangement in the HN-1 vicinity. The resulting configuration is indicated in Figure 1.1. Routine shutdown work was performed, and experiment installation work continued.

May 17 through May 22. A neutron-level decay experiment was performed. General maintenance and routine shutdown work was performed. Flux mapping of the core was performed preceeding the start-up.

June 16 through June 19. A "neutron segregation experiment" was performed to determine the effect of deuterium (γ,n) neutrons on the log N channel. Routine operational shutdown work and general maintenance were performed.

Table 1.3 lists the unscheduled shutdowns which occurred this quarter.

TABLE 1.3. UNSCHEDULED SHUTDOWNS

Date	Duration (hr)	Remarks
4-23	0.901	Gamma chamber gave scram due to low trip-point setting.
4-28	0.350	Electrical failure due to electrical storm.
4-30	0.050	Dropped No. 6 rod, reason unknown.
4-30	0.033	No. 3 safety suddenly increased to 113 N_L, reason unknown.
5-17	1.500	Setback from GCR experiment; and after recovering to full power received a setback (110 on safety recorder) which resulted in loss of rod, giving shutdown. Unable to return to power.
5-30	0.200	Resulted from disturbance on ORNL electrical
5-30	0.317	Resulted from disturbance on ORNL electrical
5-31	0.367	Resulted from disturbance on ORNL electrical
5-31	11.716	Resulted from disturbance on ORNL electrical; resulted in core reloading.
5-31	0.183	Dropped No. 4 rod, reason unknown
6-8	0.050	Faulty log N recorder
6-9	0.150	Dropped No. 4 rod, reason unknown
6-11	3.750	Dropped No. 6 rod, reason unknown-Unable to return to power due to Xenon poison.
6-12	6.233	Emergency loading continued
6-12	0.400	Faulty instrument in HN-1 instrument caused scram
6-15	0.250	Dropped No. 3 rod, reason unknown
6-23	0.183	Dropped No. 4 rod, reason unknown
6-25	0.300	Dropped No. 3 rod, Faulty magnet amplifier
6-27	0.267	Dropped No. 4 rod, reason unknown
	27.200	TOTAL

Table 1.4 gives an analysis of the causes of shutdowns for this quarter.

TABLE 1.4. ANALYSIS OF ORR SHUTDOWNS

	Number	Total Downtime (hr)
	Scheduled Shutdowns	
Regular, end of cycle	4	566.415
Regular, mid cycle	1	12.917
Research	2	4.116
Subtotal	7	583.448
	Unscheduled Shutdowns	
Instrument Failure	13	14.250
Power Failure	5	12,950
Subtotal	18	27.200
TOTAL	25	610.648

Water Systems

The modifications to the Trane air-coolers were completed. Performance tests to determine the effect of the modification were completed in May. Results indicate the units are still approx. 24% deficient. Pressure drop tests were also made to determine the effect of the new method of attaching the turbulators. Results indicate an average pressure drop across the units of 100 inches of H_2O.

Gaseous analysis of reactor water is indicated in Table 1.5.

TABLE 1.5. GASES IN ORR WATER

Gas	% of Volume
O_2	22.0
H_2	45.0
CO_2	4.8
N_2	28.2

Dissolved gas = 0.0062 cc/gm water.

Hydraulic Tube

The hydraulic tube system, which was installed April 6, 1959, has been undergoing a testing and check-out period. Sufficient performance data has been obtained to permit samples to be irradiated in this facility on a routine basis.

Beam-Hole Plug Modification

Modified beam-hole plugs, Figure 1.5, are being installed in the horizontal beam holes. To date, modified plugs have been installed in HB-1, HB-3, HB-4 and HB-5. The service piping for the beam holes to permit filling, draining, and purging is being completed as the plugs are installed.

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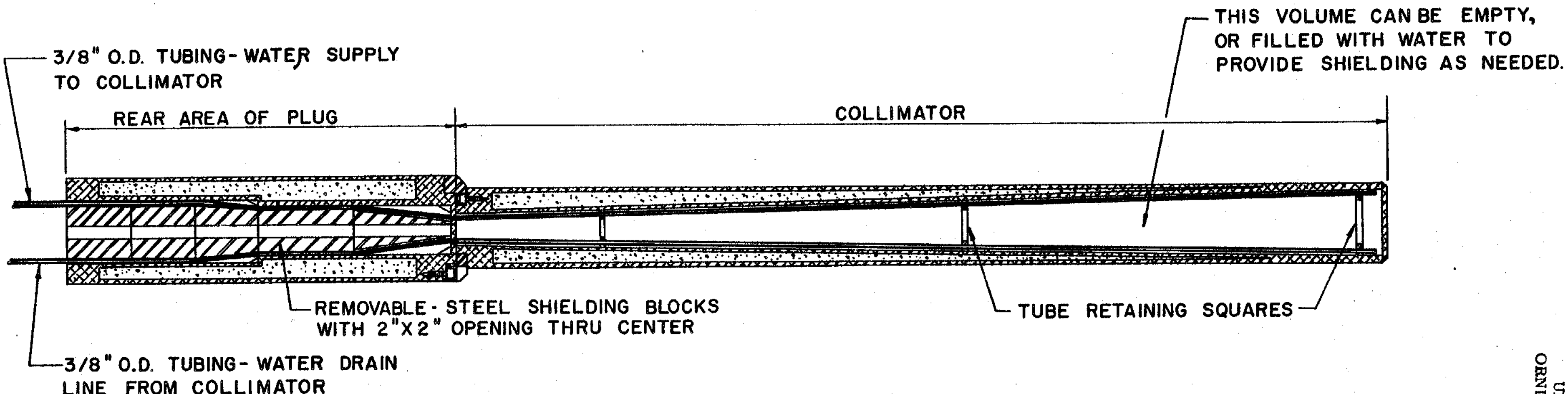

Fig. 1.5

TYPICAL SECTION THRU MODIFIED BEAM HOLE PLUG

Reactor Controls

An ORR prototype magnet was installed on No. 4 shim rod on 5-21-59 to permit study of the rod's performance under actual operating conditions. The rod has dropped a number of times since then and it is planned to investigate the mechanical adjustments of the rod during the next shutdown.

The ORR servo control system is being modified so that control shimming of a preferred shim rod will be actuated when the following conditions have been met:

(a) The servo control rod is in the withdraw limit

(b) The servo is turned on

(c) The operator is raising the servo demand

(d) The reactor is in the "run" mode (above 1% of power).

1.2. Studies of the ORR Cooling System

F. T. Binford

Some preliminary studies have been made for the purpose of determining the feasibility of increasing the power level of the ORR to 45 Mw and to determine the cooling requirements necessary at this power level.

The criteria used are as follows:

1. The maximum heat flux to be expected will not exceed 900,000 BTU/ft^2/hr. This is a straightforward extrapolation of the original design criteria employed in computing the cooling requirements at 20 and 30 Mw. While there is some reason to believe that this figure may be exceeded in positions of very high flux near the ends of the poison section of the control rods, it is felt that this problem (if it indeed does exist) can be eliminated by a minor design change in the control rods. Moreover, both the ORR boiling tests and MTR operating experience indicate that the maximum heat flux to be expected is about 820,000 BTU/ft^2/hr.

2. The maximum fuel plate surface temperature shall not exceed the saturation of water at the pressures available in the core. This value is about 240°F. Since the plate temperature must exceed this value by about 20°F at the flows contemplated to produce boiling, it is felt that the original design figure of 210°F is overly conservative.

Figure 1.6a shows the total reactor flows and water inlet temperature necessary to provide a maximum surface temperature of 240°F at 900,000 BTU/ft^{2}hr. The flow and temperature conditions should be arranged so that

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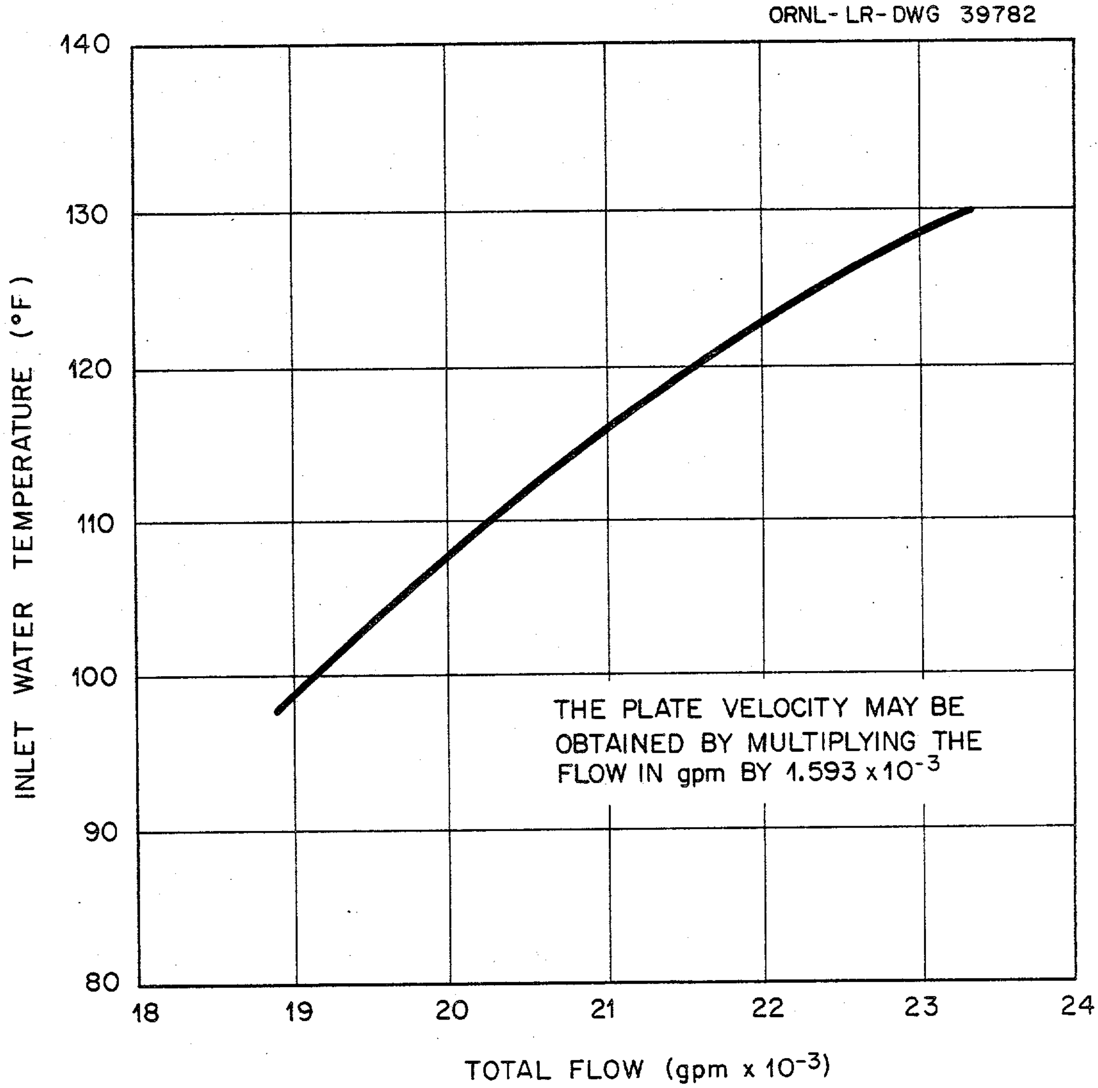

Fig. 1.6a. Total Flows and Corresponding Inlet Water Temperatures for Which ORR Fuel Plates do not Exceed 240°F at 900,000 Btu/ft^2· hr Heat Flux. (Calculated)

they fall below this curve. Figure 1.6b shows the maximum plate temperature as a function of total reactor flow under the assumption that the inlet temperature is 120°F. These are based on experimental velocities and calculated heat transfer coefficients.

The high temperatures shown in the control rod are the result of low velocities observed through one of the rods. It is believed that this can be taken care of by proper redesign so that the fuel plate temperature should be used for design purposes.

The maximum permissible flow through the reactor will depend upon the maximum pressure differential permitted on the tank. Since the pressure at the bottom of the tank is essentially the same as that in the pool, this number is proportional to the pressure drop through the core. Figure 1.7 shows the pressure drops measured across the reactor inlet and outlet lines as a function of flow. This solid line is an extrapolation of the last three points according to the relation $\Delta P = 0.074G^{2.03}$.

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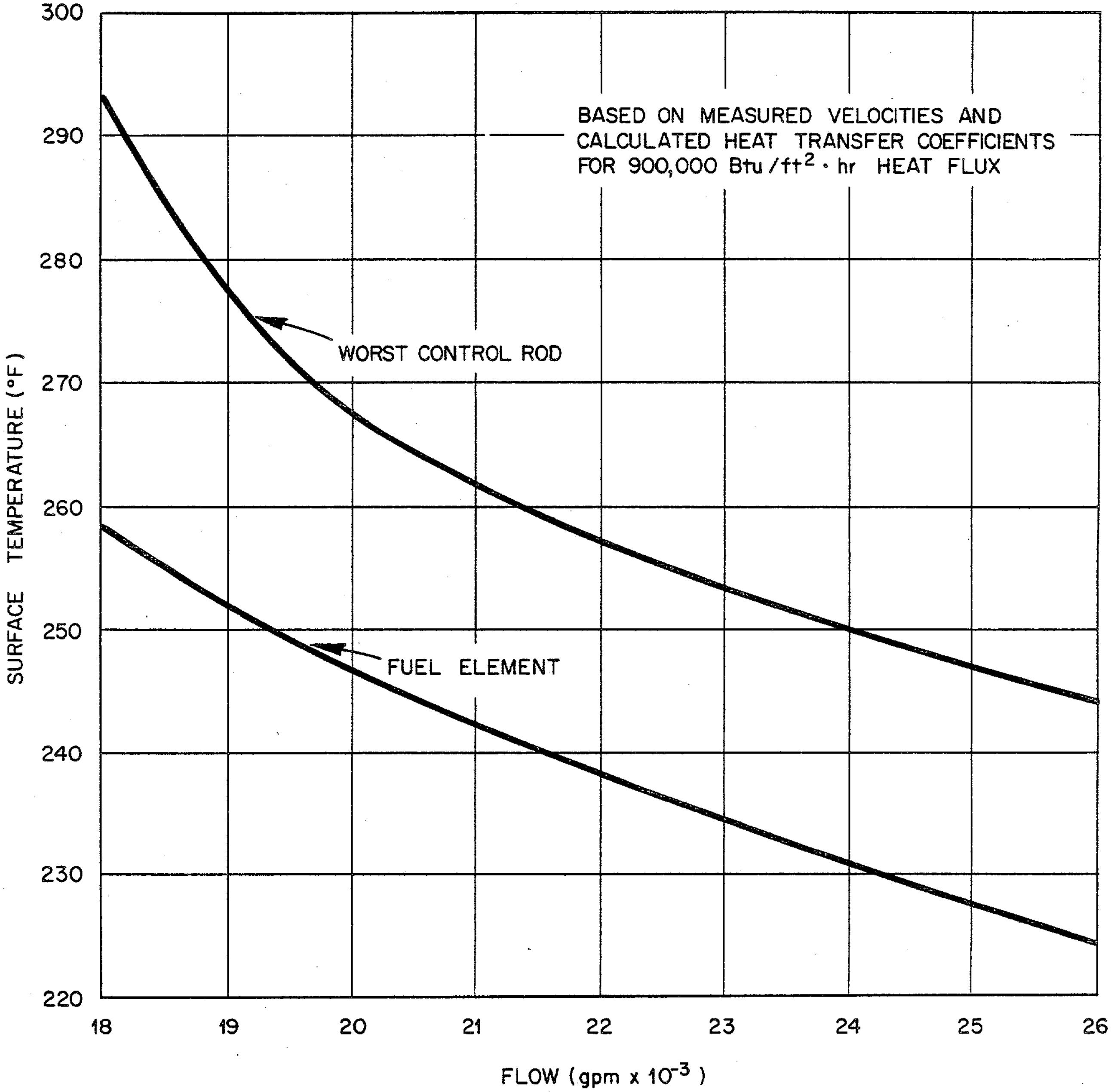

Fig. 1.6b. Maximum Plate Surface Temperature vs Coolant Flow; A 120°F Inlet Temperature.

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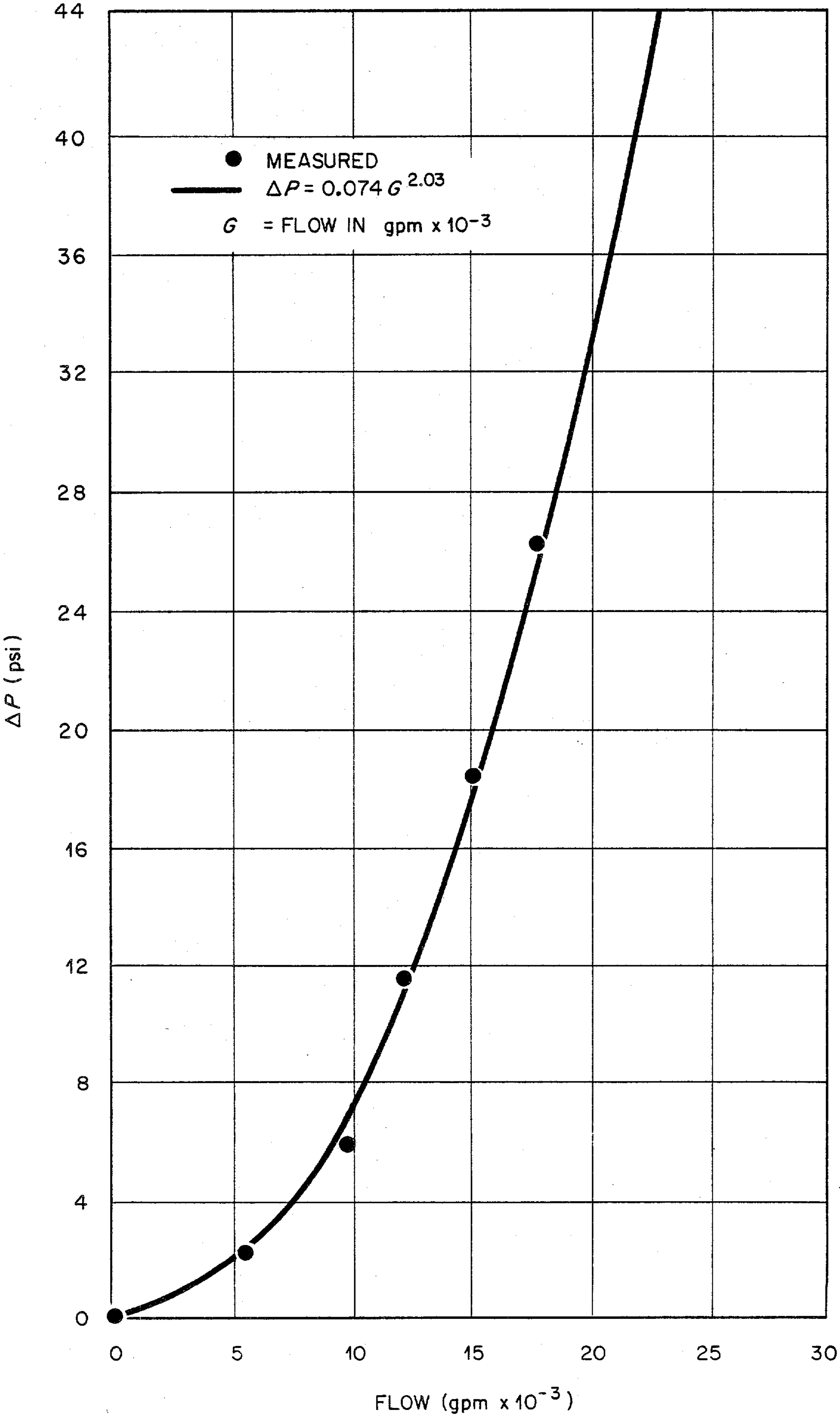

Fig. 1.7. Pressure Loss Across the ORR Tank vs Total Flow

1.3 The Effect of Photo-Neutrons on the ORR Control Chambers

A. L. Colomb

In a reactor type where the reactor and the control ionization (or fission) chambers are immersed in water, it is expected that the latter will count more and more D (γ,n) H neutrons as the distance between the reactor and the chambers is increased. Thus, it is possible to locate the chambers in a position where they will no longer give an indication proportional to the reactor neutron level. The reactor will then be controlled by an indirect measure of the gamma rays emitted by the core.

In order to check the situation at the ORR, an experiment was performed. The decay of the neutron density after a scram was measured at the three following locations:

(a) by the log N chamber at its normal location,

(b) by the reactor fission chamber positioned 19 inches from the core, and

(c) by an auxiliary fission chamber located 60 inches from the top of the core in the middle of the tank pool side.

The fission chamber in location (b) counted mainly reactor neutrons while the fission chamber in (c) saw mainly photo-neutrons.

The results of the experiment are shown in Fig. 1.8. This figure gives a plot of the apparent period read by the three chambers versus the time after the reactor scram. These results show that the log N chamber has almost the same behavior as the fission chamber which was located 60 inches from the top of the core. It is known from ORNL-2518 and from attenuation measurements that this chamber sees mainly photo-neutrons. The conclusion from this experiment is that the log N chamber is measuring mainly photo-neutrons. Further investigations are under way to determine what can be done to improve this condition.

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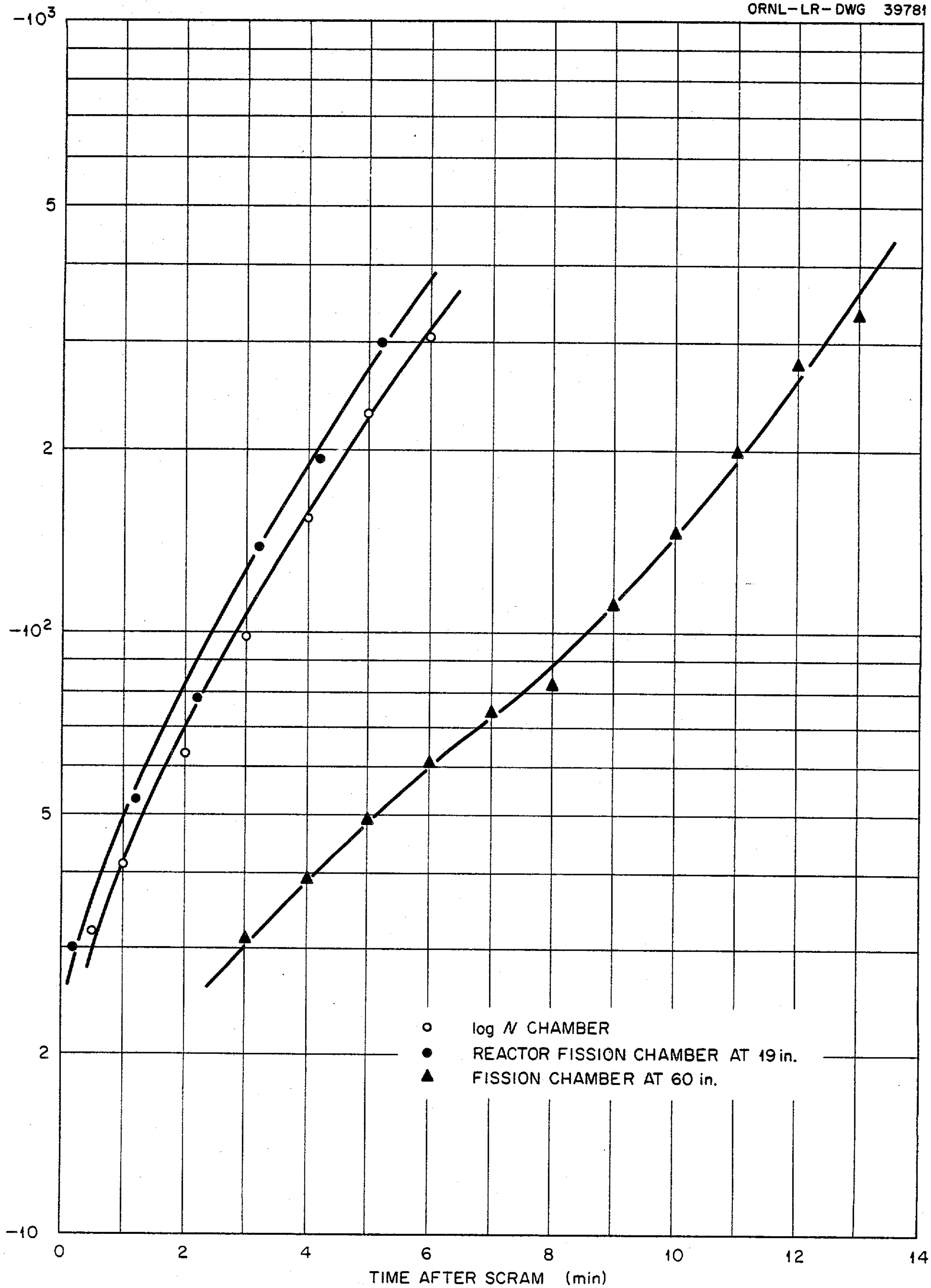

Fig. 1.8. The Effect of Photo-Neutrons on the ORR Control Chambers.

1.4 Neutron Measurements in the ORR

J. F. Wett, Jr.

Thermal neutron measurements were made in the ORR to determine the most nearly optimum core configuration for giving a high flux at HN-1. Four different configurations were measured. These loadings are shown in Figs. 1.9 through 1.12 and will be referred to as Loading 1 through 4, respectively. The method of flux measurement was the same as that used previously.[1] Flux traverses were taken in HN-1, C-5, D-1, D-9, and E-5.

Since complete core flux maps were not made, a comparison on the basis of the average core flux, as is the usual procedure, could not be made. The assumption was made that the average flux in C-5 and D-1 would not change greatly. If this assumption is true, then data could be compared to the average flux in D-1. Further, if these fluxes do not change radically, the ratio of their average should not change. It is recognized that the ratio could remain the same while the fluxes varied considerably, but this is viewed as improbable.

Data for the ratio of the average flux in a position to the average flux at D-1 and the ratio of the maximum flux in a position to the average flux in D-1 are given in Table 1.6. Data for D-9, Loading 1, and D-9, E-5, Loading 3 were lost due to improper lengths of wires being used.

1. Cagle, C.D. and Costner, R.A., Jr., "Neutron Measurements in the ORR", Operations Division Quarterly Report, October-December 1958, ORNL-CF-58-12-147.

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POOL

W

	1	2	3	4	5	6	7	8	9
A				122.63	171.74	186.81			
B			150.32	45.00	145.33	55.90	99.20		
C			112.80	150.51	134.40	149.54	163.10		195.07
D	196.42		163.49	78.78	126.67	102.04	163.58		194.66
E	196.73		168.23	146.58	135.71	141.51	163.69		194.82
F							136.38		
G									

S (left side) — N (right side)

E

TOTAL MASS: 4191.64 g U^{235}

ROD POSITIONS DURING FLUX MEASUREMENTS: 21.74 in.

Fig. 1.9. Loading No. 1.

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POOL

W

	1	2	3	4	5	6	7	8	9
A				122.63	171.74	186.81			
B			150.32	45.00	145.33	55.90	99.20		
C			112.80	150.51	134.40	149.54	163.10	195.07	
D	196.42		163.49	78.78	126.67	102.04	163.58	194.66	
E	196.73		168.23	146.58	135.71	141.51	163.69	194.82	
D							136.38		
E									

S N

E

TOTAL MASS: 4191.64 g U^{235}

ROD POSITIONS DURING FLUX MEASUREMENTS: 18.33 in.

Fig. 1.10. Loading No. 2.

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POOL

W

	1	2	3	4	5	6	7	8	9	
A				122.63	171.74	186.81				
B			150.32	45.00	145.33	55.90	99.20			
C			112.80	150.51	134.40	149.54	110.94	117.13	104.55	
S D	196.42		163.49	78.78	126.67	102.04	117.56	110.70	109.32	N
E	196.73		168.23	146.58	135.71	114.61	114.73	114.33	107.77	
F							136.38			
G										

E

TOTAL MASS: 4096.85 g U^{235}

ROD POSITIONS DURING FLUX MEASUREMENTS: 21.33 in.

Fig. 1.11. Loading No. 3.

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POOL

W

	1	2	3	4	5	6	7	8	9
A				122.63	171.74	186.81			
B			150.32	45.00	145.33	55.90	99.20		
C			112.80	150.51	134.40	149.54	110.94	117.13	104.55
D	196.42		163.49	78.78	126.67	102.04	117.56	110.70	
E	196.73		168.23	146.58	135.71	114.61	114.73	114.33	107.77
F							136.38		
G									

S (left side) — N (right side)

E

TOTAL MASS: 3987.53 g U^{235}

ROD POSITIONS DURING FLUX MEASUREMENTS: 21.20 in.

Fig. 1.12. Loading No. 4.

TABLE 1.6

Loading No.	Position	$\bar{\phi}_i / \bar{\phi}_{01}$	$\phi_i^{max} / \bar{\phi}_{01}$
1	HN-1		0.856
	C-5	2.67	3.730
	D-1	1.00	1.320
	D-9		1.440
	E-5	2.71	3.720
2	HN-1		0.980
	C-5	2.99	4.560
	D-1	1.00	1.300
	D-9	3.23	4.490
	E-5	2.97	4.360
3	HN-1		0.680
	C-5	2.67	3.800
	D-1	1.00	1.260
	D-9		
	E-5		3.730
4	HN-1		0.550
	C-5	2.39	3.640
	D-1	1.00	1.270
	D-9	1.57	2.230
	E-5	2.54	3.48

The highest ratio of maximum flux in HN-1 to average flux of D-1 is found in Loading 2. However, the ratio found in Loading 1 is close enough to warrant more consideration. Figure 1.13 shows the activity of the monitors in HN-1 normalized to the same average D-1 flux plotted against distance from the reactor. The ordinate is arbitrary. This plot gives no definite advantage to Loading 1 or Loading 2.

For a given amount of core excess reactivity at the start of a cycle, Loading 2 requires less mass than does Loading 1. Thus the average core flux in Loading 2 is higher than that in Loading 1. This gives a decided flux advantage to Loading 2. Hence, Loading 2 was decided upon as the most nearly optimum.

A complete flux mapping was made on the core for Cycle 12. This core was a slightly modified Loading 2 above and is shown in Fig. 1.14. The data for this traverse is presented in Table 1.7. Figures 1.15, 1.16, and 1.17 give some representative traverses. Figure 1.18 shows lines of equal ratio of average element flux to average core flux.

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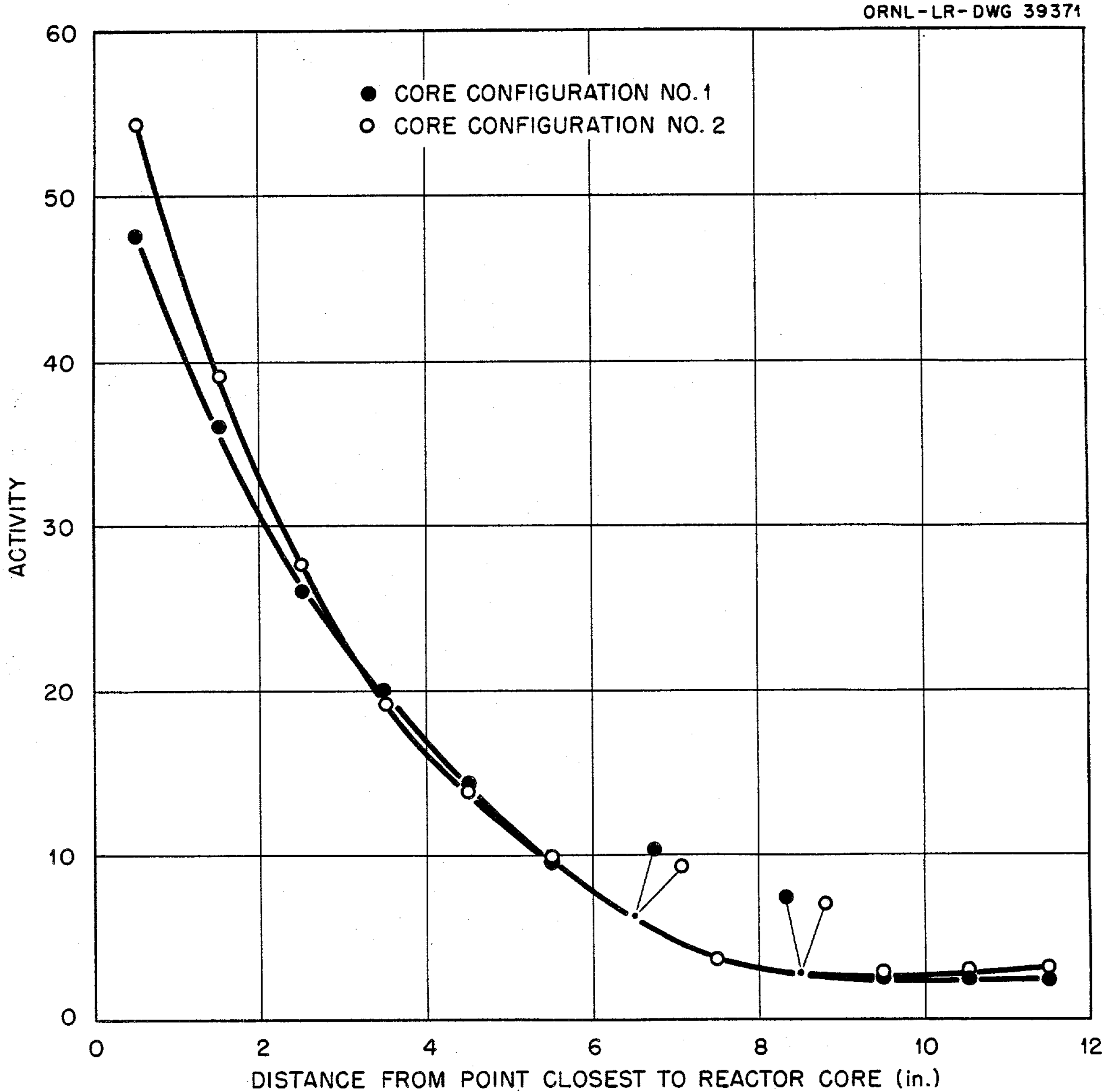

Fig. 1.13. Flux Plot at HN-1.

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POOL

W

S | N

	1	2	3	4	5	6	7	8	9
A				Pu 116.15	161.49	184.90			
B			147.25	39.25	146.84	50.15	Pu 117.00		
C			147.51	160.98	Pu 85.42	158.02	193.41	184.67	
D		187.68	144.70	65.65	119.44	87.07	161.66	182.86	
E		188.13	194.86	159.23	118.37	155.55	193.86	185.00	
F							138.64		
G									

E

TOTAL MASS: 4261.75 g U^{235}

ROD POSITIONS DURING FLUX MEASUREMENTS: 19.64 in.

Fig. 1.14. Cycle 12 a Core.

TABLE 1.7. THERMAL NEUTRON FLUX MEASUREMENTS IN CYCLE 12 CORE

Distance From Top (in.)	Flux Measurements* Core Positions			Reflector Position				
	D-5	E-4	E-5	A-8	B-2	B-8	C-2	C-9
1	0.42	0.45	0.50	0.15	0.12	0.29	0.24	0.15
2	0.33	0.44	0.43	0.21	0.18	0.31	0.28	0.24
3	0.33	0.52	0.49	0.26	0.23	0.40	0.34	0.30
4	0.39	0.59	0.58	0.32	0.28	0.48	0.39	0.35
5	0.47	0.66	0.70	0.36	0.32	0.52	0.45	0.40
6	0.58	0.77	0.78	0.40	0.36	0.60	0.52	0.44
7	0.65	0.83	0.85	0.45	0.43	0.65	0.57	0.54
8	0.75	0.90	0.99	0.49	0.48	0.73	0.68	0.58
9	0.83	1.00	1.09	0.53	0.56	0.83	0.73	0.65
10	0.95	1.05	1.19	0.61	0.60	0.90	0.80	0.71
11	1.07	1.13	1.31	0.66	0.63	1.00	0.88	0.78
12	1.20	1.22	1.39	0.73	0.73	1.09	0.99	0.84
13	1.39	1.29	1.51	0.80	0.77	1.19	1.05	0.88
14	1.53	1.31	1.56	0.85	0.85	1.24	1.14	0.91
15	1.72	1.34	1.62	0.92	0.97	1.34	1.21	0.96
16	1.75	1.34	1.63	0.95	1.02	1.38	1.26	0.97
17	1.77	1.31	1.60	0.97	1.02	1.41	1.31	0.98
18	1.73	1.29	1.55	1.00	1.09	1.50	1.34	0.99
19	1.69	1.15	1.41	1.02	1.11	1.45	1.36	1.01
20	1.56	1.05	1.34	1.02	1.14	1.48	1.38	1.02
21	1.44	......	1.22	0.99	1.11	1.41	1.34	0.96
22	1.31	......	1.11	0.95	1.07	1.34	1.28	0.91
23	1.20	......	1.07	0.92	1.02	1.23	1.22	0.86
24	1.34	......	1.36	0.83	0.97	1.19	1.16	0.79

*Expressed as the ratio of the flux to the average flux in the core which at 20 Mw was 1.10 X 10^{14} neutrons/cm^2-sec.

TABLE 1.7 (Continued)

Distance From Top (ins)	Flux Measurements* Reflector Positions D-1	D-9	E-1	E-9	F-1	F-2	G-2	G-6
1	0.15	0.79	0.19	0.23	0.09	0.18	0.15	0.19
2	0.24	0.41	0.26	0.36	0.19	0.31	0.17	0.26
3	0.30	0.54	0.32	0.45	0.25	0.37	0.22	0.34
4	0.35	0.58	0.37	0.52	0.30	0.45	0.22	0.41
5	0.41	0.66	0.43	0.60	0.34	0.51	0.31	0.50
6	0.47	0.75	0.51	0.66	0.38	0.58	0.37	0.58
7	0.55	0.85	0.56	0.77	0.45	0.66	0.43	0.61
8	0.61	0.95	0.66	0.85	0.51	0.77	0.47	0.71
9	0.65	1.07	0.75	0.95	0.58	0.85	0.52	0.77
10	0.77	1.19	0.80	1.05	0.61	0.90	0.61	0.85
11	0.82	1.29	0.87	1.16	0.70	1.00	0.65	0.92
12	0.90	1.36	0.95	1.24	0.71	1.09	0.66	1.02
13	0.95	1.50	0.99	1.31	0.77	1.16	0.77	1.09
14	0.99	1.56	1.02	1.38	0.78	1.21	0.77	1.19
15	1.02	1.60	1.09	1.39	0.80	1.24	0.80	1.29
16	1.09	1.65	1.12	1.41	0.88	1.28	0.82	1.38
17	1.12	1.67	1.16	1.44	0.88	1.31	0.85	1.41
18	1.02	1.65	1.14	1.41	0.87	1.34	0.85	1.48
19	1.09	1.65	1.16	1.44	0.88	1.28	0.88	1.50
20	1.12	1.63	1.14	1.41	0.85	1.33	0.88	1.53
21	1.07	1.53	1.09	1.34	0.80	1.31	0.85	1.50
22	1.00	1.50	1.07	1.29	0.77	1.28	0.78	1.41
23	0.95	1.36	1.00	1.19	0.75	1.21	0.77	1.33
24	0.88	1.29	0.90	1.09	0.71	1.10	0.73	1.22

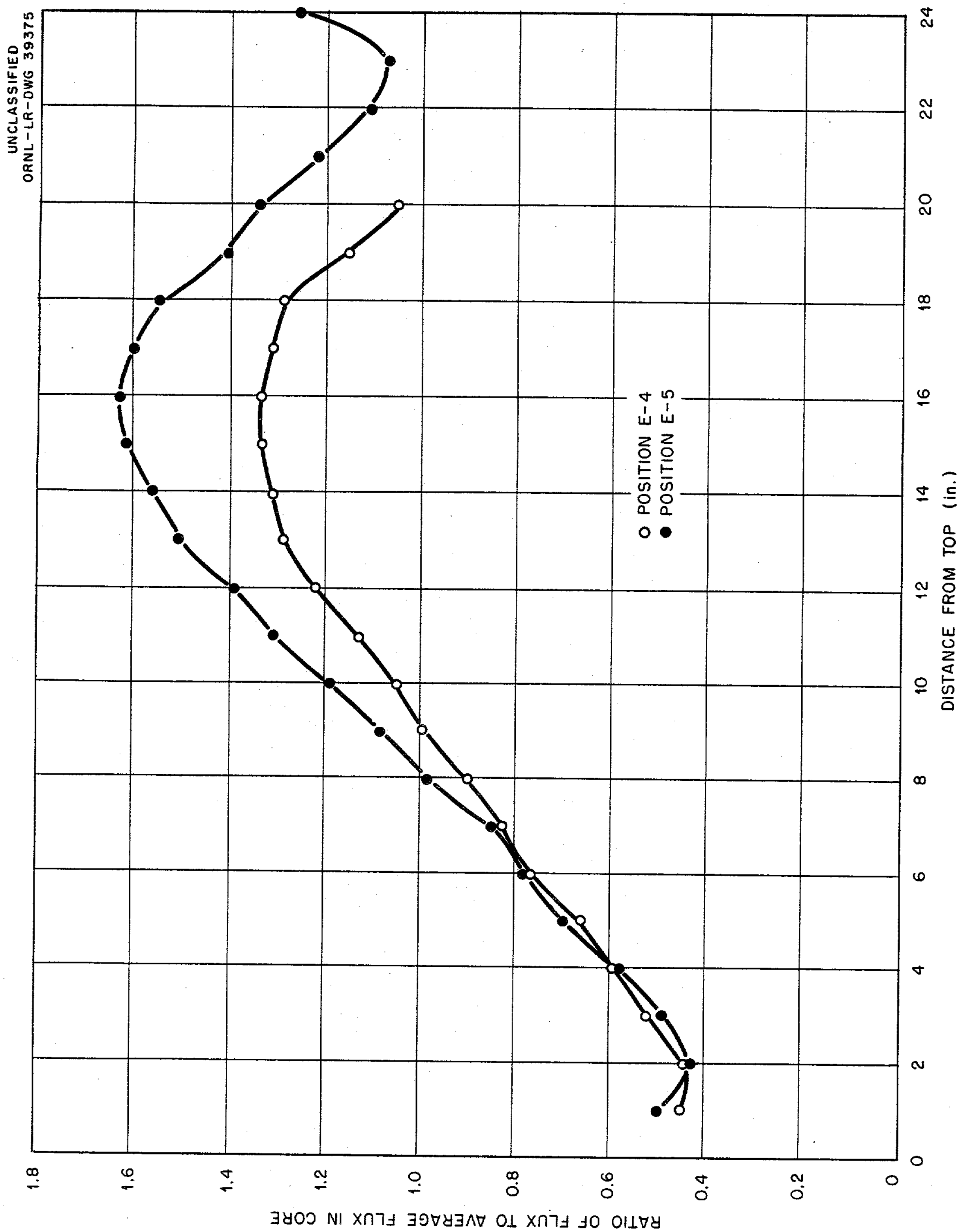

Fig. 1.15. Sample Flux Traverse.

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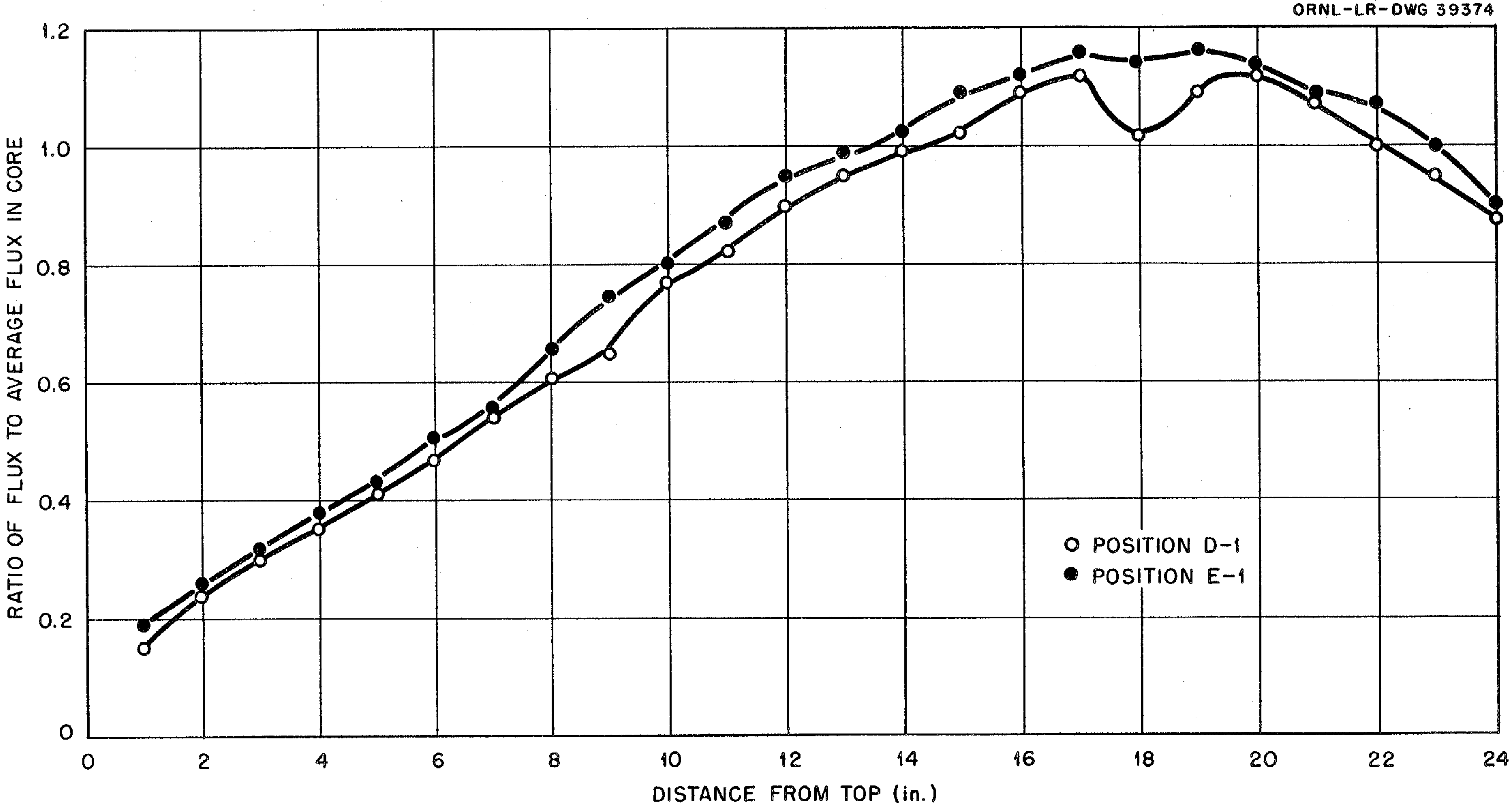

Fig. 1.16. Sample Flux Traverse.

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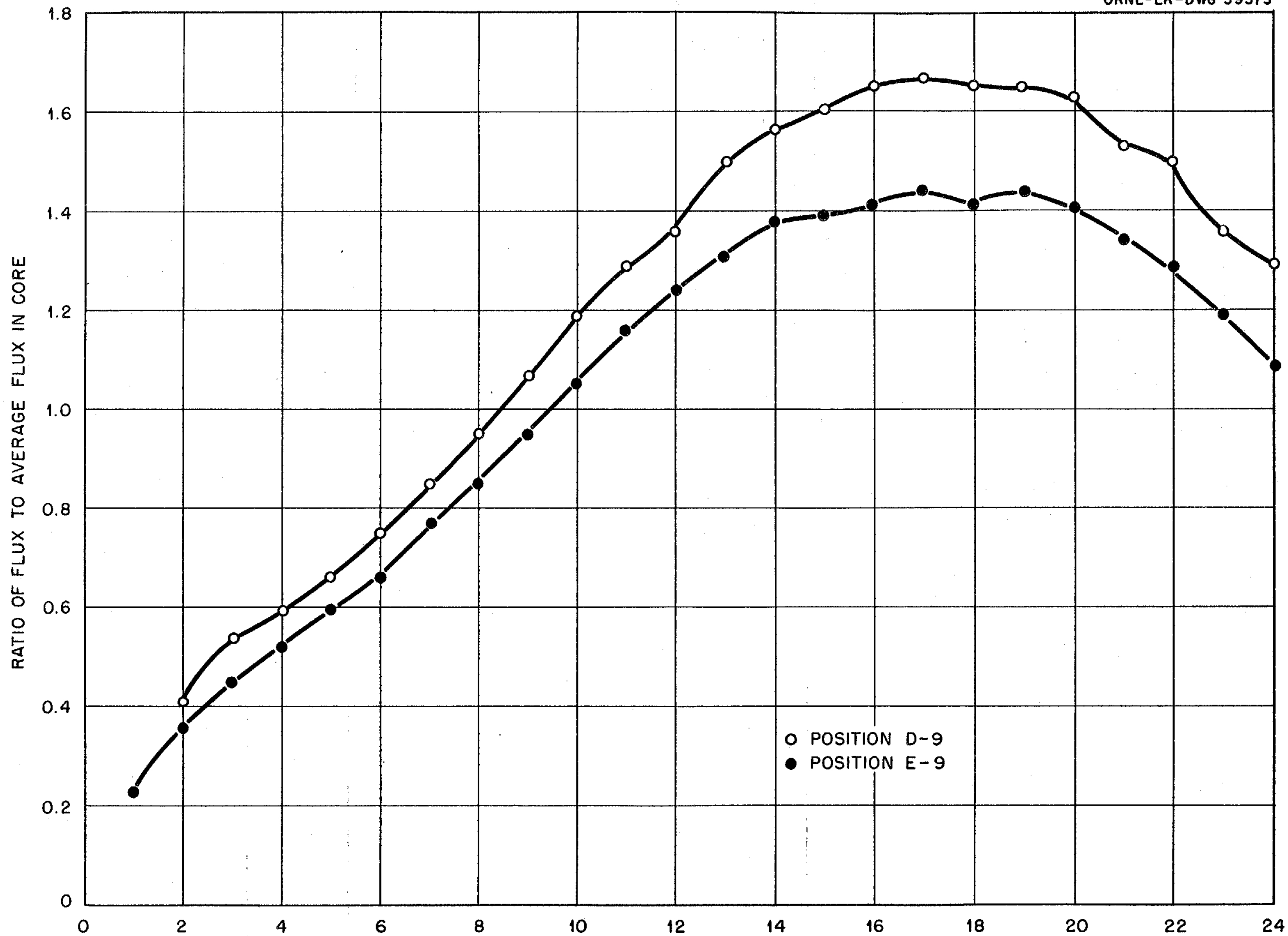

Fig. 1.17. Sample Flux Traverse.

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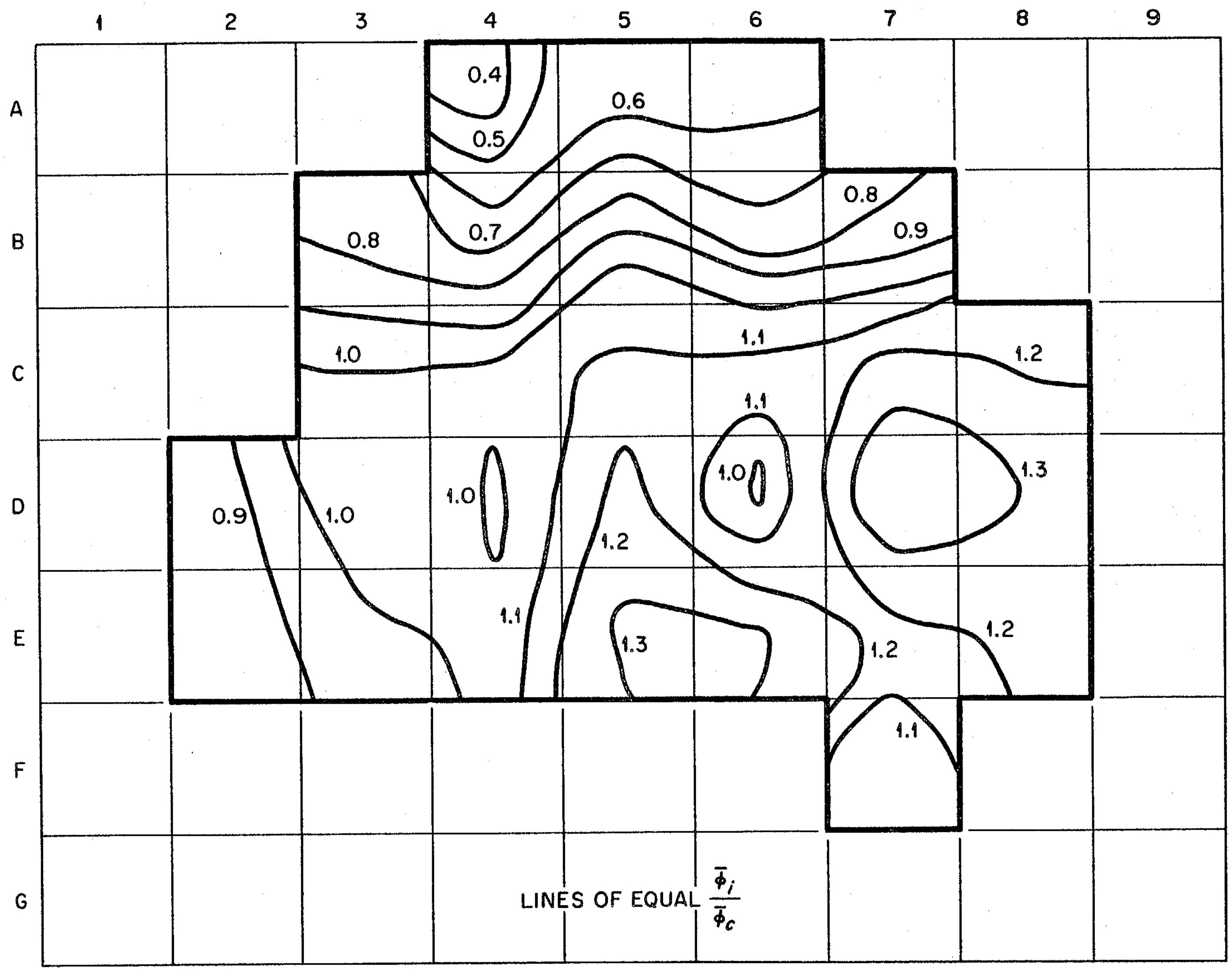

Fig. 1.18. Lines of Equal Ratio of Average Element Flux to Average Core Flux.

2. ORNL GRAPHITE REACTOR

2.1 Operations

W. R. Casto

Operations

The operating data for the ORNL Graphite Reactor are given in Table 2.1.

TABLE 2.1 GRAPHITE REACTOR OPERATIONS

Period April 1, 1959, through June 30, 1959

	This Quarter	Last Quarter	Year to Date
Total energy, Mwd	275.1	281.1	556.2
Average power, kw/operating hour	3417	3438	3428
Time operating, %	88.5	89.0	88.75
Exit air filters, Δp, in. H_2O (av)	3.88	3.95	3.91
Canal water radioactivity, c/m/ml (av)	578	721	650
Research samples	280	276	556
Radioisotope samples	300	402	702

Drilling New Experiment Hole

Experimental drilling was started on March 23, 1959, to determine if it was feasible to provide more research or operational facilities on top of the reactor.

The present hole is being drilled approximately 4 1/2 ft east of scanner holes 8 and 9 and is centered between vertical fuel rows 8 and 9.

Drilling through the 7 ft concrete shield has been completed. This is an 8-in. diameter hole through the outer 4 ft and a 6-in. diameter hole through the remaining three ft. A 4 5/8-in. diameter hole has been drilled approximately 16 in. into the graphite. Progress to date has been slow due, mainly, to difficulties encountered in drilling through a steel box full of No. 2 lead shot. This was finally accomplished by grouting the shot and drilling through after the grout had set. All drilling is being done during the Monday shutdown of the reactor. It is expected that the total depth of 21 ft will be reached in the near future.

Slug Ruptures

Slug rupture data are given in Table 2.2.

Table 2.2 SLUG RUPTURES

Rupture Number	Date	Row No.	Number From West End of Row	Lot	Remarks
257	4-20-59	1775	20	138	Found by visual inspection
258	4-27-59	1574	20	136	Found by visual inspection
259	5-4-59	1861	19	119	Found by visual inspection
260	5-21-59	1372	23	118	Indicated by probe
261	6-8-59	2270	10	109	Found by visual inspection
262	6-8-59	1766	15	110	Found by visual inspection
263	6-22-59	1376	16	150	Found by visual inspection

Raising the Canal Walls

The radiation background in the canal had become a very serious problem as it limited the working time in that area. The background in the walkway varied from 15 to 25 mr/hr, and over the wall the variation was 35 to 100 mr/hr. This high radiation background was due to Co^{60} and fission products which had soaked into the porous concrete walls along the top of the water level.

Preliminary work was begun February 2, 1959, to raise the walls an additional 18" with a stainless steel liner on the inside to give an impervious surface at, and below, the water level.

The raising of the walls was completed late in June, and the water level was raised on June 29. The radiation was successfully reduced. This is now generally 3 - 5 mr/hr or less in the normal working areas.

Radiation and Contamination in the Exit-Air Duct

To determine the amount of radiation and contamination in the exit-air duct, a removable concrete slab was removed from the exit-air duct on May 18, 1959. This slab was located approximately 50 ft from the reactor.

The reactor was shut down at 4:00 a.m. At approximately 11:00 a.m., the radiation was 4 r/hr against the concrete duct wall; and the background was generally 3 r/hr inside the duct. At 3:00 p.m. the background radiation had decreased to 2 r/hr. A wet towel smear of the wall showed 40 mr/hr contamination.

The bottom side of the removable concrete slab was decontaminated from 4 r/hr to 1 r/hr.

On this same day, a plug was removed from the drain line of the exit-air duct. This plug was approximately 6 ft under water in the canal deep pit, and it was necessary to lower the water level to remove the plug. The plug was not replaced, and during the night the fan suction pulled water from the canal into the air duct. It was necessary to limit the fan suction until the drain could be plugged.

Filter Changes

One half of the inlet air filters to the reactor were changed on 5-25-59, the balance were changed on 6-1-59.

The No. 1 filters in No. 2 cell of the filter house were changed on 6-1-59.

Reactivity Test

A standard procedure has been established for obtaining the relative reactivity value of slug rows in which fuel changed. Critical tests are made before and after discharging the normal 4-in. bonded type slugs which are being replaced by SAR and hollow type slugs. (The hollow slug is referred to as a dingot* slug).

The 4-in. bonded slugs contain 1170.65 grams of normal uranium, the SAR solid slug contains 1939.1 grams of normal uranium, and the dingot slug contains 1786.24 grams of normal uranium.

Table 2.3 gives the information collected.

---

* The dimensions of the MARK VII A Dingot slugs are as follows:
OD 1.19 - 1.21 in.
ID 0.49 - 0.51 in.
Al thickness 0.042 - 0.043

TABLE 2.3 REACTIVITY DATA

Date	Channel	Slug Row 4-inch	Inhours Before Discharging 4-inch Slugs	Inhours After Discharging 4-inch Slugs	Inhours Lost	Inhours After Recharging With SAR Slugs	Slug Row 8-inch	Inhours Gained
3-23	1963	51	---	336.0	---	352.5	26	16.5
4-27	1574	51	---	365.0	---	377.0	25	12.0
6-1	0568	54	321.0	322.0	+1.0	Did not recharge		
6-1	1068	54	334.0	321.0	13	Did not recharge		
6-8	2270	41	336.0	326.5	9.5	Did not recharge		
6-8	1766	41	326.5	339.5	13.0	Did not recharge		
6-22	1376	54	293.0	280.5	12.5	292.5*	27*	12.0*

*Recharged with Dingot slugs.

2.2 Graphite Reactor Annealing

L. E. Stanford

The Safeguard Report on the Proposed Method of Annealing Graphite in the X-10 Reactor was completed and submitted to the AEC for approval on April 20, 1959. It appears that approval will not be forthcoming in time to permit the annealing to be done during the current summer, and this is now expected to take place during the summer of 1960.

The accumulation of graphite temperature data is continuing. Approximately 60 thermocouples have been installed in the reactor and are presently being used to obtain temperature traverses through several fuel channels. Representative data for an inlet air temperature of 86°F are shown in Fig. 2.1 for full power operation, and Fig. 2.2 shows similar data for the same fuel channels at half power operation (1.75 Mw). The maximum fuel element temperature was adjusted to 240°C for both the full-power and half-power run by adjusting the coolant-flow rate. These data show that a considerable increase in graphite temperature, relative to the fuel element surface temperature can be obtained by operating the reactor at reduced power levels and air flows.

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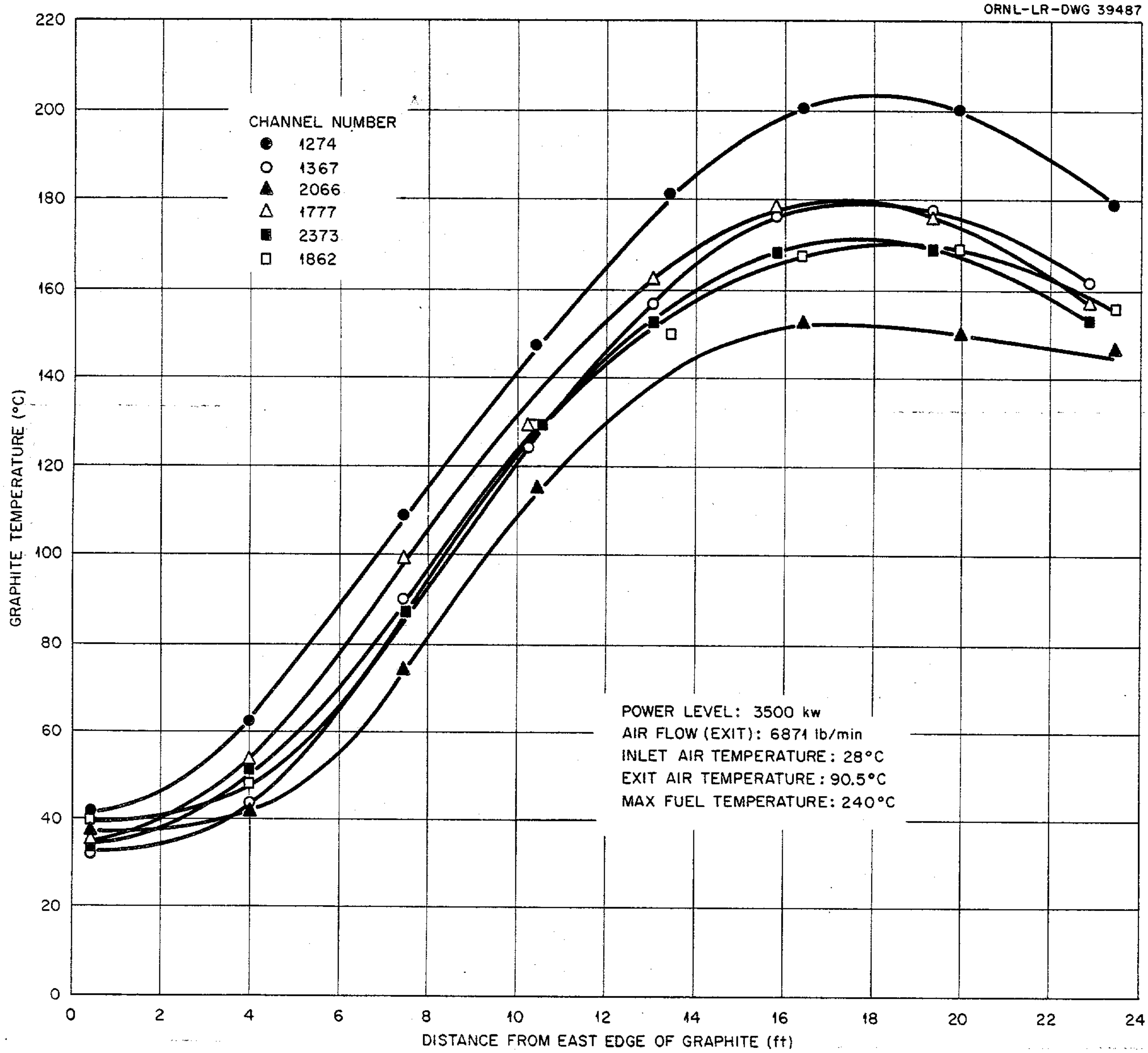

Fig. 2.1. Graphite Temperature Distributions Through Unloaded Fuel Channels (Channel Air Flow Blocked) for Summer Operation at Full Power-Full Flow.

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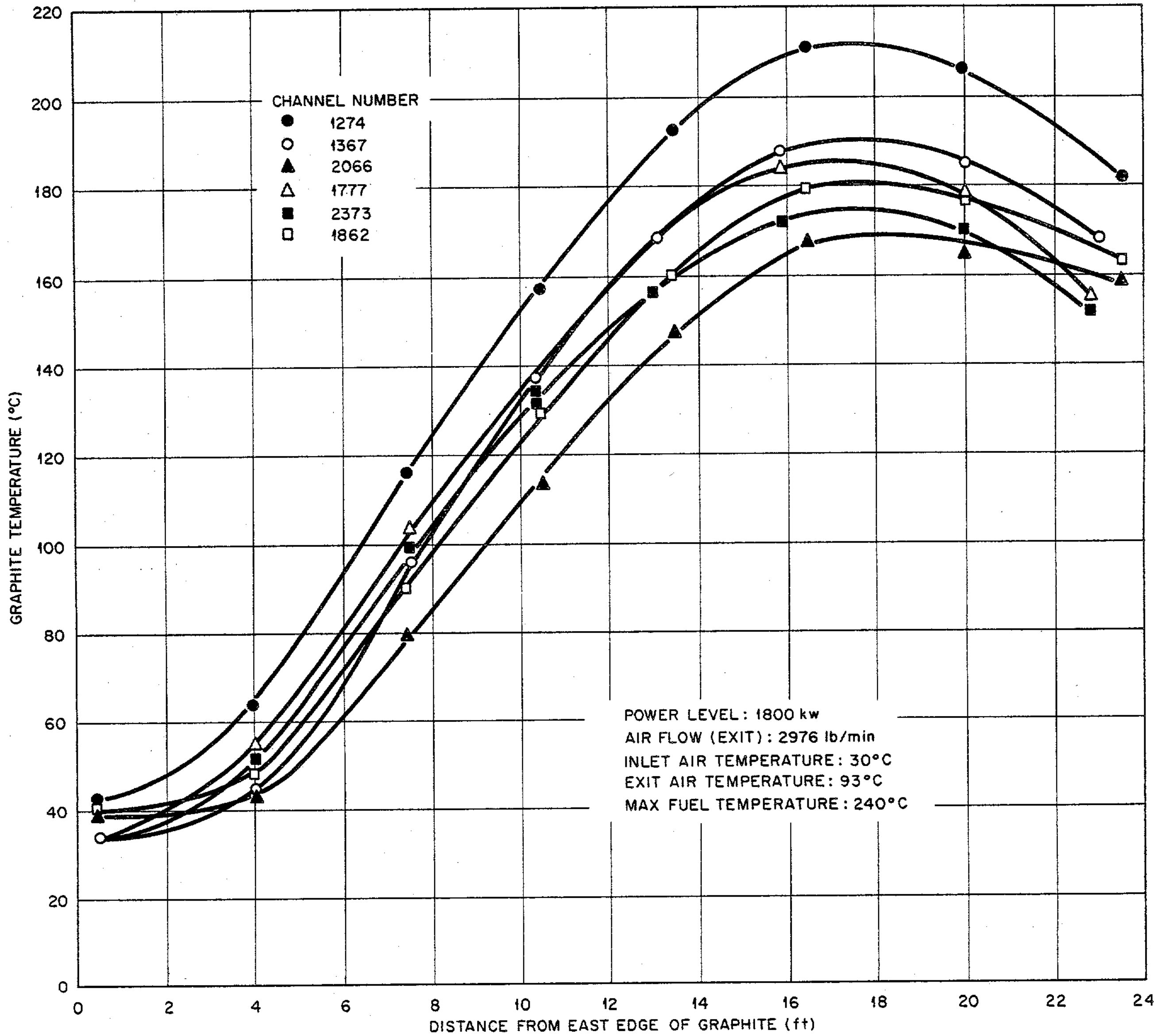

Fig. 2.2. Graphite Temperature Distributions Through Unloaded Fuel Channels (Channel Air Flow Blocked) for Summer Operation at Half Power-Half Flow.

3. LOW-INTENSITY TEST REACTOR

3.1 Operations

W. R. Casto

Operations

The operating data for the LITR are given in Table 3.1.

TABLE 3.1 LITR OPERATIONS

Period April 1, 1959 through June 30, 1959

	This Quarter	Last Quarter	Year to Date
Total energy, Mwd	238.2	227.4	465.6
Average power, kw/operating hour	2,973	2,971	2,972
Time operating, %	88.0	85.0	86.5
Cooling water radioactivity, c/m/ml	49,800	45,000	47,400
Cooling water resistivity, ohm-cm	897,000	836,000	866,000
Fuel pieces charged	6*	3	9
Fuel pieces discharged	6*	3	9
Shim rods charged	1	1	2
Shim rods discharged	1	1	2

* 3 partial fuel elements

Experimental Facilities

As evidenced by Figure 3.1 there has been no change this quarter; however, several changes are under way to take effect next quarter, and they will be reported at that time.

Radioactivity in Cooling Water

Table 3.2 gives some of the principal radioactivities found in LITR cooling water samples during the quarter.

TABLE 3.2 RADIOACTIVITIES IN LITR COOLING WATER

	dis/min/ml						
Date	I^{133}	I^{131}	Np^{239}	Na^{24}	Cu^{64}	Ti^{99m}	
6-23-59	1.76×10^3	184	934	2.3×10^4			
5-13-59	1.4×10^3	200	975	2.08×10^4	2.5×10^3	3×10^3	
4-23-59		187	785	1.71×10^4			Si<0.5ppm

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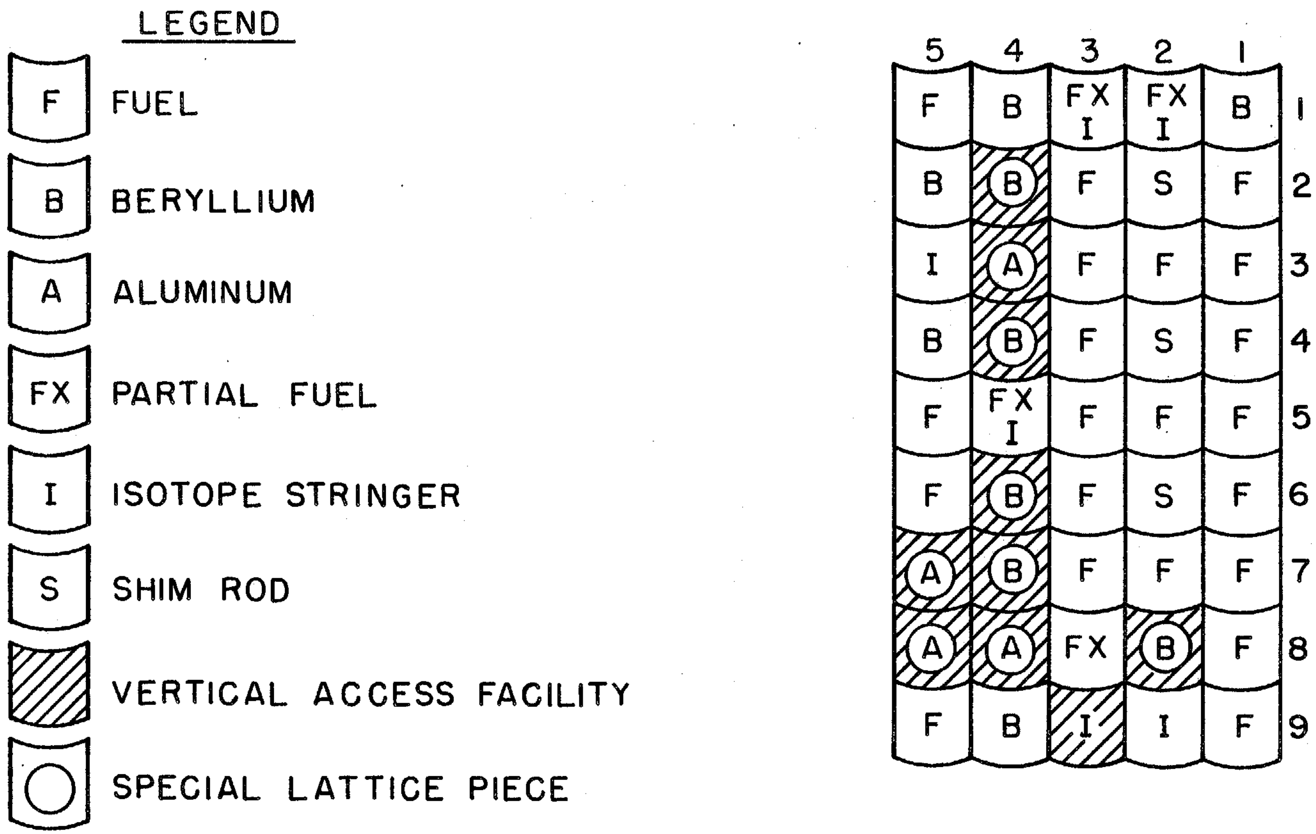

Fig. 3.1.

LITR LATTICE CONFIGURATION

2ND QUARTER, 1959

TABLE 3.3 LITR LATTICE INVENTORY AS OF JUNE 30

Position	Material	Weight (g)	Position	Material	Weight (g)	Position	Material	Weight (g)	Position	Material	Weight (g)	Position	Material	Weight (g)
51	F	125.389	41	Be		31	Fx	46.691	21	Fx	94.830	11	Be	
52	Be		42	Be (SP)		32	F	196.950	22	F (SR)	123.064	12	F	199.940
53	Is		43	Al (SP)		33	F	153.423	23	F	177.908	13	F	146.930
54	Be		44	Be (SP)		34	F	159.065	24	F (SR)	113.506	14	F	137.283
55	F	126.831	45	Fx	92.170	35	F	154.946	25	F	177.626	15	F	144.423
56	F	155.061	46	Be (SP)		36	F	167.121	26	F (SR)	97.640	16	F	186.641
57	Al (SP)		47	Be (SP)		37	F	138.085	27	F	165.232	17	F	156.734
58	Al (SP)		48	Al (SP)		38	Fx	133.044	28	Be (SP)		18	F	156.324
59	F	144.017	49	Be		39	Is		29	Is		19	F	163.813

Legend:
F - Fuel
Be - Beryllium
Fx - Partial Fuel Element
SR - Shim Rod
SP - Special Piece
Is - Isotope

3.2 LITR Shim-Safety Rod Checks

C. D. Cagle

In an effort to increase the worth of the No. 1 rod, the LITR shim-safety rods have been arranged so that the newest rod is in the No. 1 position and the oldest is in the No. 3 position. New fuel additions for burnup compensation were concentrated around the No. 1 position. A check of the period produced versus rod withdrawal distance showed that the No. 2 and No. 3 rods are roughly equal in reactivity worth and that the No. 1 rod is now worth about 15% less than the No. 2 rod.

4. LABORATORY FACILITIES

E. J. Witkowski

4.1 Radioactive Waste Disposal

White Oak Creek Discharge to Clinch River

The Health Physics monitoring data for radioactive waste discharged to the Clinch River during the last four quarters are listed in Table 4.1. The highest weekly discharge of 159% of MPC during the last quarter resulted from a low river flow which was only 8% of average.

TABLE 4.1 CLINCH RIVER MONITORING DATA

	% of MPC* in Clinch River	
	Average for Quarter	Highest Weekly Discharge
April 1 - June 30, 1959	44	159
January 1 - March 31, 1959	84	683
October 1 - December 31, 1958	48	400
July 1 - September 30, 1958	80	240

* The MPC in the Clinch River is the weighted average of the MPC values for occupational exposure of the individual radioisotopes as set forth by national and international committees on radiation protection. For prolonged exposure of a large population, it is recommended that the permissible levels be reduced by a factor of ten.

Discharges to White Oak Creek and Lagoons

A comparison of discharges based on gross beta analysis for the quarter, year to date, and the year 1958 is given in Table 4.2. Although the gross beta analysis under normal conditions is only sufficiently accurate for operating control purposes, it was far more inaccurate during this quarter because it failed to pick up the large amount of Promethium-147 that was accidentally discharged into the system. As indicated in Table 4.3, approximately 70 curies of Promethium-147 that are not shown in the gross beta analysis were discharged to White Oak Creek from the Process Waste Treatment Plant.

There were two accidental releases of Promethium-147 in the Isotope Area. One resulted from a spill in the radioisotope storage barricade and the other from decontamination of the vacuum lines. The drain in the barricade is to be rerouted to the "hot" waste system to prevent similar occurrences in the future.

TABLE 4.2 DISCHARGES TO WHITE OAK CREEK AND LAGOONS

	This Quarter	Year to Date	Total for 1958
Process waste discharged to White Oak Creek, gal	80,330,000	151,810,000	233,340,000
Activity discharged to White Oak Creek, curies	38.5	57.7	91.8*
Waste to lagoons, gal	807,000	1,721,000	3,157,000
Activity to lagoons, curies	60,042	72,095	52,795
Total activity discharged to lagoons 2, 3 and 4 to date, curies	239,081		
Total volume discharged to lagoons 2, 3 and 4 to date, gal	13,424,000		

* Lowest annual discharge on record

Process Waste Treatment Plant

The operating rate was increased from 300 gpm to 330 gpm at the end of the quarter. Analytical results are not yet available for use in determining the decontamination factors at the increased flow rate.

The main activity processed through the plant was the Promethium-147 accidentally released in the Isotope Area.

TABLE 4.3 PROCESS WASTE TREATMENT PLANT OPERATING DATA

Contaminant	Activity Curies In	Activity Curies Out	Activity Curies Removed	Percent Decontamination
Sr^{90}	26.37	6.48	19.89	76
Sr^{89}	1.38	0.29	1.09	79
Ru^{106}	2.53	0.51	2.02	80
Cs^{137}	30.69	5.60	25.09	82
Co^{60}	2.62	0.38	2.24	86
Pm^{147}	862.00	69.12	792.88	92
Total rare earths	90.71	9.99	80.72	89

Gas Disposal

The monitoring data for stacks 3020 and 3039 are given in Table 4.4.

A new steam line to the turbines at 3039 stack, independent of the Isotope Area operations, was installed to improve the reliability of the emergency fans and blowers and to provide for expansion of the off-gas facilities.

Cell ventilation and off-gas services were extended to the former Waste Evaporator, Building 3506, now being used for radioisotope production.

The Cottrell precipitator at the 3039 stack was shut down for 23 hours for emergency replacement of three insulators which had dropped off and made the equipment inoperable. During the shutdown, an hourly check was maintained on the stack discharge; it was not necessary to discontinue any of the Laboratory's operation to prevent a serious discharge from the stack.

A valve was installed in the stack drain to eliminate the backflow of stack gases through the chemical waste drain lines to the Isotope Area buildings. The condition was created by the build-up of pressure at the base of the stack with the addition of new blowers in recent years.

TABLE 4.4 STACK MONITORING DATA

	% of Maximum Permissible Operating Level*	
	Stack 3020	Stack 3039
Average daily discharge	5	0.9
Highest daily average	13	17

*The maximum permissible operating level for air contamination has been arbitrarily established by Health Physics so as not to exceed more than 10% of the permissible exposure. The stack discharges are restricted to values which, in combined effect and under the worst conditions of dispersion, would not exceed at any point ground concentrations equal to the maximum permissible operating level.

4.2 Manipulator Cell Operations

Dismantling Cells in Building 3026

The CPFF contractor has completed the work deleted from the lump sum contract and the refurbishing of the inside of the building. Installation of equipment has started. The operating personnel has been selected and are being trained to operate the facility.

The installation of equipment has been delayed for removal and replacement of paint in the zinc bromide windows. The repainting is approximately 90% complete. The air operated hoists in cells B-1 and B-2 were installed, and the installation of the tunnel transfer cart is approximately 90% complete.

Operating personnel are working with the Chemical Technology Division group connected with the PRFR Mechanical Phase Program, becoming familiar with the special equipment that is to be installed later in the Dismantling cells.

Several other groups in the Laboratory have shown renewed interest in the cells now that they are nearing completion.

Cells at the ORR

These cells have been operated very informally by the Laboratory Facilities Department, with the aid of the Reactor Operations Department, while a more formalized procedure could be developed. Numerous alterations to the cells have been initiated to make them more operable and safer. Solid States, Physics, Chemical Technology, and Reactor Chemistry-G.E. have been the main users of the cells.

A questionnaire, to be filled out by the prospective users of the cells, is being developed along with operating procedures. Before starting a new operation, the completed questionnaire will be reviewed by the Laboratory Facilities Department and the Technical Department whose primary interest will be in safety with respect to occupants and operations of the ORR.

Alterations to the cells that have been completed or that are now in progress include the installation of a cat-walk with hand rails on top of the cells; installation of an eight-inch lead brick wall now separating the two cells; fabrication of lead access doors for each cell; installation of an intercommunication system in each cell; and the installation of a hot off-gas system in the south cell with provisions for a connection in the north cell.

4.3. Liquid Hydrogen Dispensing Facility

The construction of the facility is complete with the exception of several alterations that are necessary to insure safer operations. The changes in equipment, as well as extensive changes in the first written operating procedures, were suggested by the Bureau of Standards Cryogenic Laboratory's experts with many years of experience in handling liquid hydrogen. If no further delays are encountered, the facility should be ready to receive liquid hydrogen for dispensing sometime in August 1959.

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